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Study of the Allegro Core Performance

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ABSTRACT

The presented paper is related to introduction of the design and neutronic characterization of the start-up core developed for Gas cooled Fast Reactor (GFR) demonstrator. Slovak University of Technology in Bratislava joined the project ALLEGRO in last decade within the consortium of middle-European institutions. In the development plan of the GEN IV GFR the ALLEGRO demonstrator is one of the most necessary steps. The ALLEGRO reactor is small helium cooled 75 MW_{th} thermal power unit. Its main objective is to demonstrate the key GFR technologies and to perform tests of innovative materials. The reactor core is based on the standard and MOX pin type fuel in the first phase of the project. The active core of a large GFR 2400 makes use of ceramic materials, but in the first ALLEGRO core MOX fuel will be used. It will be mainly to demonstrate the viability of the technology and to acquire necessary experimental data for further research. In the presented works are identified the main discrepancies between ALLEGRO and GFR 2400 designs, the sensitivity analysis was performed for both reactors. Neutronic characterization is aimed to determination of the standard neutronic parameters using conventional computational systems. The results of sensitivity and uncertainty calculations are presented in conjunction with similarity analysis.

Keywords: *Neutronics, MOX type fuel, Code Validation, GFR demonstrator*

INTRODUCTION

The Gas-cooled Fast Reactor (hereinafter the GFR) is one of the six most promising reactor concepts selected by the Generation IV International Forum (GIF) [1]. The design of this reactor may partially benefit from previously proposed but not realized conceptions of the Sodium-cooled Fast Reactor (SFR) and the Very High Temperature Reactor (VHTR). Despite the extensive research done into the GFR technology, no gas cooled fast reactor has ever been built. In the development plan of the GEN IV GFR the ALLEGRO demonstrator is one of the most necessary steps. The ALLEGRO reactor is small helium cooled 75 MW_{th} thermal power unit. Its main objective is to demonstrate the key GFR technologies and to perform tests of innovative materials. The active core of a large GFR 2400 makes use of ceramic materials, but in the first ALLEGRO core MOX fuel will be used. It will be mainly to demonstrate the viability of the technology and to acquire necessary experimental data for further research. In order to identify the main discrepancies between ALLEGRO and GFR 2400 designs, the sensitivity analysis was performed for both reactors. The obtained results predict approach how to design and optimize the ALLEGRO core for testing of experimental assemblies. ALLEGRO core is characterized by standard mixed oxide (MOX) assemblies consisting of fuel pins with stainless steel cladding operated at an average coolant temperature around 400°C. In contrast, GFR 2400 core is based on carbide pin fuel type with the application of refractory metallic liners used to enhance the fission product retention of the SiC cladding.

The main principles of fast reactor systems are rather well understood, however, their optimization, in order to comply more effectively with requirements and their timely deployment, requires the research in nuclear data. Although most nuclear data are by and large available in modern data files, their accuracy and validation is still a major concern. The main source of uncertainty in the calculated core responses is due to uncertainties in evaluated nuclear data such as microscopic cross sections (XS), fission spectra, neutron yields, and scattering distributions that are contained in cross section evaluations. These uncertainties are governed by probability distributions which are unknown, but the evaluated data values are assumed to represent the mean of the distribution. TSUNAMI-IP utility available in SCALE system [2] computes the contribution to the response uncertainty due to the cross-section covariance data with the use of sensitivity profiles of the investigated system. The results of this uncertainty analysis of ALLEGRO MOX core are given and discussed in this paper. In addition the neutronic similarity of ALLEGRO MOX core to the several hundred critical benchmark experiments specified in the ICSBEP Handbook [3] is evaluated by the use of three integral indices.

The ALLEGRO reactor is expected to be built in the central European region, thus the research of this reactor is the driving mechanism for Slovak institutions to participant in research projects and in the development of GEN IV fast reactors.

1 GENERAL DESIGNS OVERVIEW

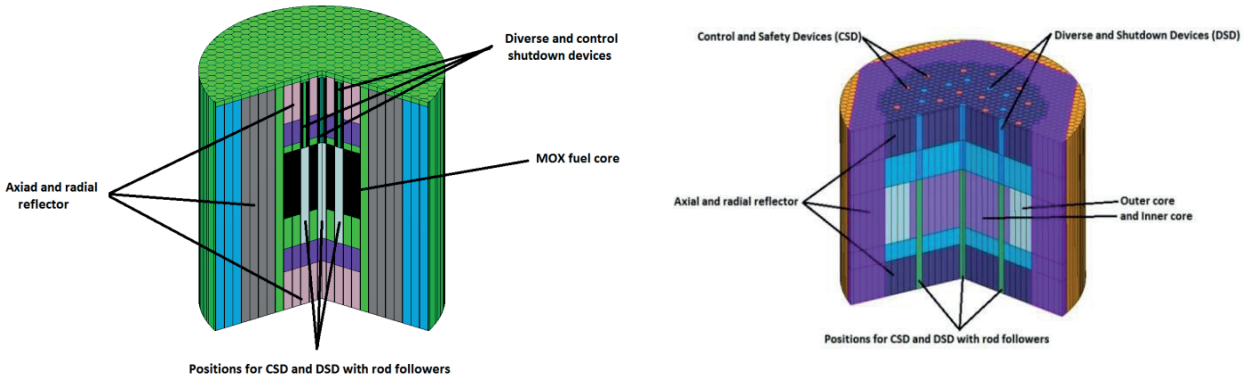
1.1 ALLEGRO core specification

The ESNII+ ALLEGRO MOX starting core configuration is an experimental unit with thermal power of 75 MW_{th} characterized by standard Mixed OXide (MOX) fuel assemblies consisting of fuel pins with stainless steel cladding operating at an average helium coolant temperature around 400 °C. The 120 degree symmetric core includes 81 fuel assemblies, with 169 fuel pins. The average PuO_2 content of the heavy metal material in the fuel pin is 25.5%vol. In addition, the ALLEGRO MOX core features 6 in-core dummy assemblies made of special steel alloy 15-15Ti (AIM1) so far assumed homogeneous in geometry and composition. The control rod system is composed of 4 Diverse Shutdown Devices (DSD) and 6 Control and Shutdown Devices (CSD). The absorber rods in both groups are composed of boron carbide. The core fuel region is surrounded by four additional rings of reflector assemblies (80% vol. AIM1 + 20% vol. He at 70 bar) in the radial direction and by the 30.2 cm high axial reflectors of almost the same material placed above and below the fission gas plenums. Additional three rings of shielding assemblies are placed around the reflector in the radial direction. Axial shielding placed just under and above the axial reflector region uses a lower amount of B_4C. In order to ensure adequate heat transfer the primary coolant pressure during normal operation is 7 MPa. Three decay heat removal loops with helium-gas heat exchangers are available to mitigate core melting accidents. The global primary arrangement is based on two main helium-water loops (2x38 MW_{th}), each fitted with one intermediate heat exchanger (IHX) blower unit. The produced heat is finally transferred by air coolers from the secondary circuit to the atmosphere as the ultimate heat sink.

1.2 GFR 2400 Core Specification

The GFR 2400 reactor is considered as a conceptual design of a large scale power GFR. This design is based on the foregoing concepts and experiences of all GoFastR [4] participants. In European FP7 programme, the GoFastR project was the Euratom contribution to the Gen IV gas cooled fast reactors (GFR). The GFR 2400 design is a large scale power unit with thermal power of 2400 MW_{th}. This fast-spectrum reactor is a helium-cooled system and it works with a closed fuel cycle. Primary coolant pressure during normal operation reaches 7 MPa in order to ensure adequate heat transfer. Due to safety reasons, the coolant volume fraction in a core is high. This fact allows

maintaining the natural circulation of coolant under pressurized conditions even if active systems are not available. Three decay heat removal loops (each 100% capacity) with heat exchangers and forced convection devices are available and six additional gas reservoirs are prepared in case of emergency to mitigate core melting accident. The global primary arrangement is based on three main loops (3 × 800 MW_{th}), each fitted with one IHX blower unit, enclosed in a single vessel. The current choice of power conversion system is the indirect Brayton cycle operated with He-N_2 mixture. The planned cycle efficiency is approximately 45%. Since the GFR 2400 components need to withstand high temperatures, ceramic compositions are under investigation as a promising solution for used materials. The pin type fuel of active length of 165cm consists of uranium plutonium carbide (UPuC) and it is surrounded by tungsten-rhenium compound (W14Re) and rhenium (Re) refractory liners to ensure fission products confinement within the pins. The gap between fuel and liners is filled with helium gas of 1MPa pressure. The use of a SiC_f/SiC material for fuel cladding is the latest and very promising solution, where the SiC fibers are improving the mechanical properties of the fuel pin. The active core consists of two zones. The PuC volumetric content in inner core fuel assemblies (FA) reaches 14.2%, and 17.6% in the outer core. The isotopic composition of uranium corresponds to natural abundance of its isotopes while plutonium composes of the twice recycled MOX fuel, expected to be available in France from 2016 [5]. The core fuel region is surrounded by six rings of Zr_3Si_2 reflector assemblies in the radial direction and by the 1m high axial reflectors of the same material placed above and below the fission gas plenums. The general views of ALLEGRO MOX and GFR 2400 core designs can be found in Figure 1.



a) ALLEGRO MOX b) GFR 2400

Figure 1: 3D cross-sectional views of ALLEGRO MOX and GFR 2400 core designs

2 CALCULATION METHODOLOGY

The sensitivity and uncertainty analysis of ALLEGRO MOX and GFR 2400 cores were performed by using two computational tools. In the first case, the TSUNAMI-3D code was utilized using ENDF/B-VII [7] 238 group cross section data and 44groupcov covariances. Forward and adjoint transport calculations were carried out with KENO6 and the sensitivity coefficients were computed by the SAMS module. For the neutron flux calculations square mesh was placed through the core with a uniform step of 1.5 cm in the fuel region. In other parts of the core, the size of the mesh was directly proportional to the distance from the core centre. In the second case, self-developed perturbation PORK code was used which is interconnected with the diffusion flux solver DIF3D [6] and ZZ-KAFAX-E70 [8] based ENDF/B-VII nuclear data library collapsed from 150 to 25 groups. Due to the multi-group cross section data used in both computational routes, in a process of cross section preparation the resonance self-shielding calculation had to be performed. SCALE system is capable to perform only cell calculation at the level of fuel pin with definition of cladding and coolant in an infinite lattice, but with an option where the spectral calculation can be carried out by using CENTRUM code. Methods used in the multi-group cross sections processing procedures

for DIF3D calculation allow us to take into account resonance self-shielding effect as well as the spatial boundary effects. In this case, two level of cross section calculation was necessary to perform, where in the second level the cross sections were condensed from 150 to 25 groups structure by using regional-wise neutron flux from RZ transport calculation. Sensitivity coefficients calculated by TSUNAMI-3D were collapsed to 25 group structure.

TSUNAMI-IP utility uses sensitivity data generated by TSUNAMI-1D and/or TSUNAMI-3D sequences and cross section-covariance data stored in the 44GRPCOV library. TSUNAMI-1D/3D are sequences that execute modules to determine response sensitivities and uncertainties. The linked computations perform the cross section self-shielding operations, forward and transport calculations, computation of sensitivity coefficients and calculation of the response uncertainty. The SCALE covariance library is based on several different uncertainty approximations with varying degrees of fidelity to the actual nuclear data evaluation. The library includes evaluated covariances obtained from ENDF/B-VII, ENDF/B-VI, and JENDL3.3 for more than 50 materials. It is assumed that covariances taken from one data evaluation such as ENDF/B-VI or JENDL-3.3, can also be applied to other evaluations of the same data, such as ENDF/B-VII. If this is done judiciously for cases in which the nuclear data evaluations are similar, then the covariances taken from one source should be a reasonable representation of uncertainties for the other evaluations. ORNL has a database of pre-calculated sensitivity profiles for several hundred critical benchmark experiments specified in the ICSBEP Handbook. These sensitivities may be input to TSUNAMI-IP utility, along with calculated sensitivity profile of application system. In our case 494 benchmark experiments with various energy group structures were used.

Three global integral indices [2] are used in the analysis to assess the similarity of ALLEGRO MOX neutronic core design (hereinafter application – index a) and a single experiment (e) on a system-wide basis for all nuclides and reactions. Each integral index is normalized such that a value of 1.0 represents complete similarity between ALLEGRO MOX core design and specific benchmark experiment and the value of 0.0 indicates no similarity. The uncertainty of the integral response ΔR (for instance k_{eff}) on the target integral parameter by the use of XS sensitivity coefficients denoted by symbol S and XS covariance matrix M can be evaluated by the well-known sandwich formula:

$$\Delta R^2 = S_R M S_R^T , \tag{1}$$

where the impact of the individual reactions and energy groups can be evaluated separately. The diagonal elements of the resulting matrix, defined as the solution of Eq. (1), represent the relative variance values for each of the system under consideration. The off-diagonal elements are the relative covariances between given experiments. Following the SCALE methodology, these covariances transformed to correlation coefficients (*ck*) describe the degree of correlation (coupling) in the uncertainties between the two specific systems. This correlation (coupling) demonstrates the level of similarity in the predicted response biases between various systems in the frame of XS induced uncertainties. The E parameter given by Eq. (2) assesses similarity between two systems based on the magnitude and shape of all sensitivity profiles.

$$E = S_a S_e^T / \left| S_a \right\| S_e \right| . \tag{2}$$

If the group-wise sensitivity data for all nuclides and reactions for each system are considered as a vector, the index E is the cosine of the angle between the two sensitivity vectors. If theses vectors are parallel (E=1), the systems are proportional. The G index assesses the similarity of two systems based on normalized differences in the energy dependent sensitivity data for fission, capture and scatter. A physical interpretation of the G index is the ratio of the sum of the sensitivity

coefficients of the application that are covered by the experiment to the sum of the sensitivity coefficients of a given application. The *G* index is defined as follows:

$$G = 1 - \sum_{n}\sum_{x}\sum_{j}\left(S_{x,j}^{a,n} - S_{x,j}^{e',n}\right) \Big/ \sum_{n}\sum_{x}\sum_{j}\left(S_{x,j}^{a,n}\right), \tag{3}$$

where the symbol *n* stands for the number of application system nuclides, *x* represents the reaction and *j* the summation which is performed over all energy groups. As it can be seen from Eq. (3), a *G* value of 1 indicates complete similarity and a *G* value of 0 indicates no similarity. The nuclide-reaction specific partial integral index based on the same coverage criteria as *G* is denoted *g*.

3 RESULTS

Mentioned above, both investigated systems (ALLEGRO starting core and GFR 2400) are characterized by different material composition. ALLEGRO is composed from MOX fuel with steel type cladding and reflector. Although the GFR 2400 uses carbide fuel with SiC cladding and zirconium based reflector, these differences may not necessarily result in a totally different performance of both cores initiated by a same event. First, the spatially averaged neutron spectra were compared in Figure 2. In the energy region above 1 MeV, the shape and magnitude of the neutron flux is almost identical. Maximum of the spectrum observable below this energy is more significant for ALLEGRO core. In the case of GFR 2400, neutrons from this energy area are moved to lower energies probably due to slowing down on carbon nuclides. This effect can be seen also in the energy region between 1keV and 50keV, where the magnitude of GFR 2400 flux spectra is higher compare to ALLEGRO and the maximal value is slightly higher than the maximum which lies near 500 keV energy. A valuable neutron population in resonance energies is the result of competitive interactions of neutrons with carbon and not only with uranium. Material with similar properties like carbon is missing in the core composition of ALLEGRO MOX core therefore the neutron spectra is harder.

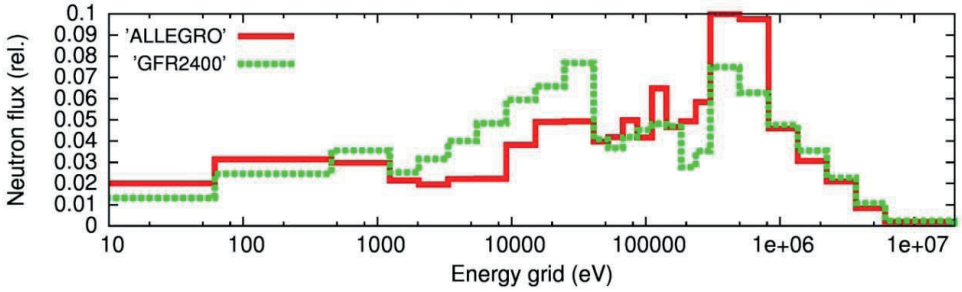
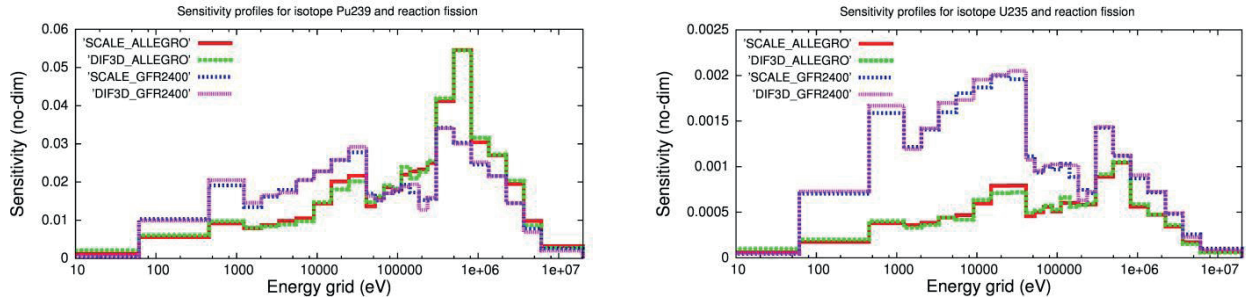


Figure 2: Normalized neutron flux spectrum from DIF3D calculation

The first group of analyzed nuclides consists of fissile nuclides. In this case, the composition of fuel vector for both systems was the same. The difference was just in plutonium isotopes enrichment. The good agreement between the shape of sensitivity profiles and neutron flux can be seen on the sensitivity profiles of ^{239}Pu and ^{235}U, which are presented in Figure 3-a and b. In the case of Pu, the shape of sensitivity profiles is consistent with Figure 2. A little different case is U, where the shape of GFR 2400 sensitivity profile is comparable to neutron flux spectrum but it is intensified by increasing value of ^{235}U fission cross section towards lower energies which ultimately results in notable overestimation compare to ALLEGRO MOX sensitivity profile. From the global view to fissile nuclides, the multiplication properties of both systems are very similar, because they are led by ^{239}Pu (almost equal integral sensitivity coefficient of fission), but with different consequences during transients. In case of a power excursion where the fuel temperature increases, neutron absorption in the resonance region becomes dominant reaction due to Doppler broadening

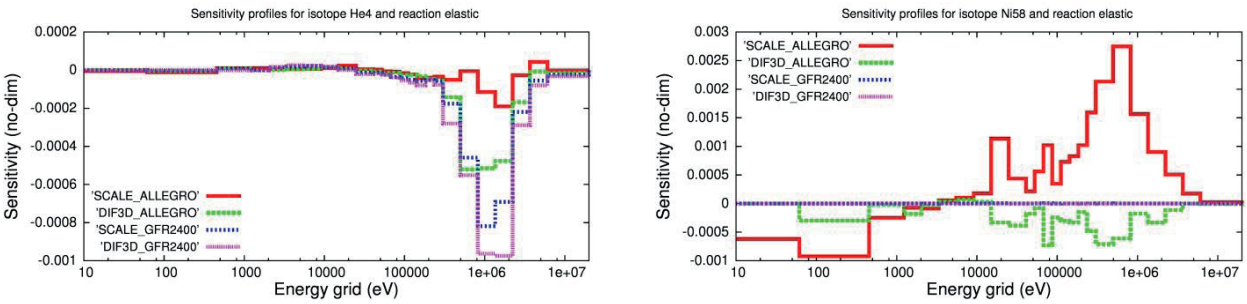
which will result in a more negative temperature reactivity effect for GFR 2400 core compared to ALLEGRO MOX core.



a) b)

Figure 3: Sensitivity profiles of fissile nuclides and reaction fission

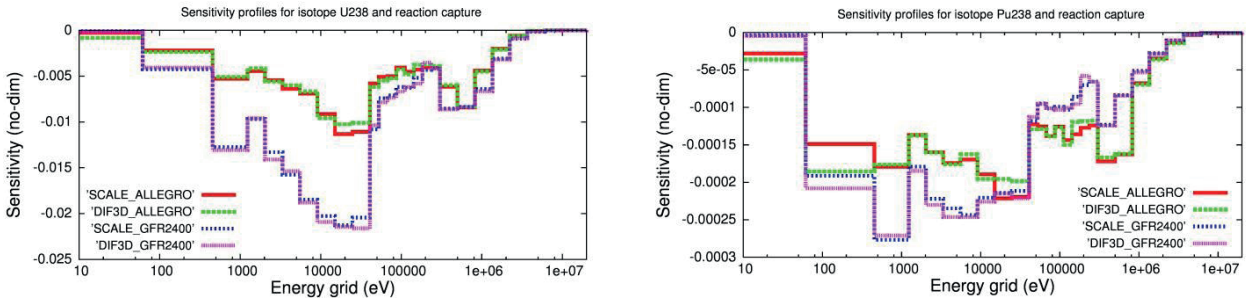
A special attention within sensitivity analysis was paid to structural materials including also a reflector material and coolant. For this group of isotopes it is impossible to expect some comparable results because each core is based on different sets of materials but their individual contribution to the multiplication properties of corresponding system may identify unique processes hidden in integral parameters. Comparison of sensitivity profiles of ^{4}He and elastic scattering, presented in Figure 4-a, identifies discrepancy between these profiles both at the level of comparison between the systems, as well as within comparison between used code schemes. The main inconsistency is in case of ALLEGRO MOX core where the shape of sensitivity profile calculated using DIF3D code scheme (green dot line) is comparable with the sensitivity profiles of GFR 2400 core but the sensitivity profiles calculated using SCALE system (red line) is depressed close to zero value which makes ALLEGRO MOX core almost insensitive to change in He composition. Main consequence of this inconsistency could be improper estimation of void effect by SCALE system or overestimation of the same effect in DIF3D. In a certain way, similar behavior was noticed for structural materials of ALLEGRO MOX core as Fe, Cr or Ni, and elastic scattering reaction. GFR 2400 core contains a minimal amounts of these isotopes therefore sensitivity profiles presented in Figure 4-b are almost zero. The sensitivity profiles of ^{58}Ni and elastic scattering for ALLEGRO MOX core, shown in Figure 4-b, illustrate a different response to change in elastic scattering cross section based on computational scheme. In case of SCALE system, sensitivity profile is mainly positive and peak-oriented around energy 600 keV with small negative contribution in lower energies. However, the sensitivity profile calculated by DIF3D scheme is purely negative. Shape of both profiles can be considered with some simplifications as a comparable but the main inconsistency is related to an absolute magnitude of individual sensitivity coefficients. The source of discrepancies presented in Figure 4-a and b are related to diffusion solution used in DIF3D. Information about non-symmetric angular distribution of elastic scattering related to border effect near fuel-reflector interface is probably lost by using scalar flux during sensitivity calculation. Within comparison of sensitivity profiles between systems, contribution of for GFR 2400 total sensitivity is almost ten times higher than in case of ALLEGRO which is slightly surprising due to absence of materials with moderator properties. This effect can be explained by an effective contribution of fission in lower energies of GFR 2400 core.



a) b)

Figure 4: Sensitivity profiles of chosen nuclides and elastic scattering reaction

The last set of sensitivity profiles is focused on capture reaction. In Figure 5, results for ^{238}U and ^{238}Pu are presented. In the case of ^{238}U, based on the high sensitivity in resonance energy region for GFR 2400 in connection with sensitivity of ^{239}Pu fission, stronger temperature reactivity effect can be expected compared to ALLEGRO MOX core. Based on this different sensitivity in the resonance region, extrapolation of Doppler coefficients determined for ALLEGRO MOX core to GFR 2400 will be incorrect.



a) b)

Figure 5: Sensitivity profiles of chosen nuclides and reaction capture

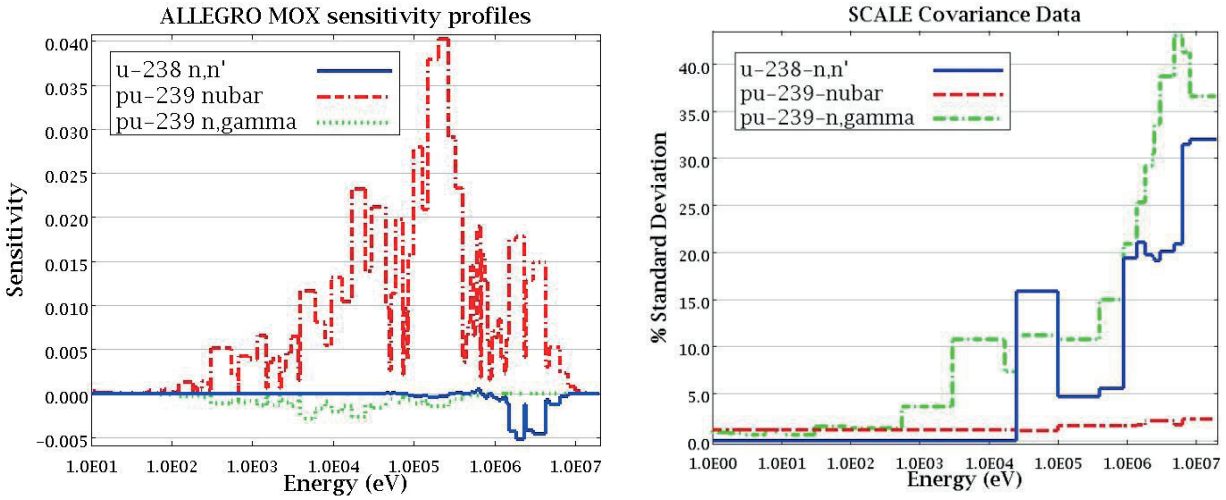
As it was mentioned above, the TSUNAMI sequence computes the contributors to the application response uncertainty due to the XS covariance data. The relative standard deviation of ALLEGRO MOX k_{eff} due to XS covariance data is 1.04%. Table 1 lists the top 16 covariance matrices that contribute to the k_{eff} uncertainty. These contributors represent more than 98% of the total uncertainty induced by XS data. The k_{eff} in the case of 238 energy group calculation with control and safety rods reaches 1.02534 ± 0.00019.

Table 1: Uncertainty contribution in ALLEGRO MOX k_{eff}

No.	Covariance Matrix		Contributions to Uncertainty in k_{eff} (% Δk/k)	No.	Covariance Matrix		Contributions to Uncertainty in k_{eff} (% Δk/k)
	Nuclide-Reaction	Nuclide-Reaction	Due to the Matrix		Nuclide-Reaction	Nuclide-Reaction	Due to the Matrix
1	^{239}Pu nubar	^{239}Pu nubar	6.7999E-01	9	^{238}U n,gamma	^{238}U n,gamma	1.5155E-01
2	^{238}U n,n'	^{238}U n,n'	5.0948E-01	10	^{238}U nubar	^{238}U nubar	1.1712E-01
3	^{240}Pu nubar	^{240}Pu nubar	2.3377E-01	11	^{56}Fe elastic	^{56}Fe elastic	9.6235E-02
4	^{239}Pu n,gamma	^{239}Pu n,gamma	2.3310E-01	12	^{56}Fe n,gamma	^{56}Fe n,gamma	7.5133E-02
5	^{239}Pu chi	^{239}Pu chi	2.1225E-01	13	^{241}Pu fission	^{241}Pu fission	6.7164E-02
6	^{238}Pu fission	^{238}Pu fission	2.0489E-01	14	^{240}Pu fission	^{239}Pu fission	5.8365E-02
7	^{238}U elastic	^{238}U n,n'	1.9741E-01	15	^{52}Cr elastic	^{52}Cr elastic	5.4093E-02
8	^{239}Pu fission	^{239}Pu fission	1.8240E-01	16	^{239}Pu n,n'	^{239}Pu n,n'	5.4073E-02

The top contributor to k_{eff} uncertainty is the ^{239}Pu nubar value. This is due to the large PuO_2 volume fraction (25.5%) in the MOX fuel and, as can be seen in Figure 6, also due to the high sensitivities above 100 keV threshold. In case of ^{238}U n,n' reaction there are large negative sensitivities in the energy range above 1 MeV burdened with significant relative standard deviation of XS data (20 - 35%). Although the k_{eff} sensitivities to ^{239}Pu n,gamma reaction are in magnitude much smaller than ^{238}U n,n' and ^{239}Pu nubar, the uncertainty associated to XS data is large and varies between 5 to 45% in the relevant energy range. The similarity assessment procedure identified three groups of potential experiments, where the values of the *ck* coefficients got over 0.4.

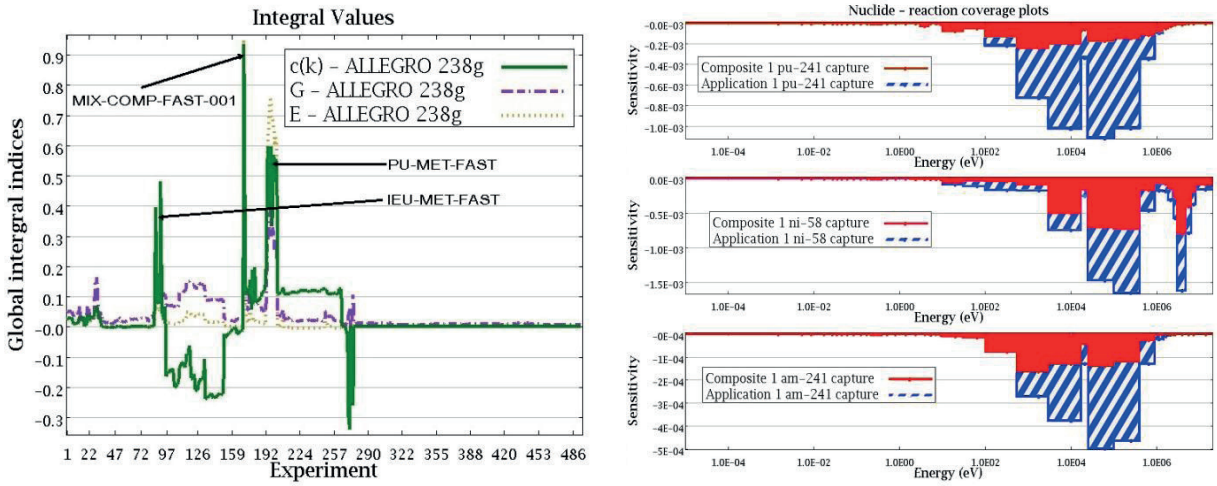
However, as it can be seen in Figure 7-a), only one experiment (MIX-COMP-FAST-001-001) [3] reached *ck* greater than 0.9.



a) b)

Figure 6: Application sensitivity profiles and covariance data

The good similarity results are mainly driven by the type of fuel (MOX) and fuel cladding material used in the ALLEGRO MOX model and in the MIX-COMP-FAST-001-001 experiment. Although the *E* coefficient reaches the quite high value (0.95), the big portion (25%) of ALLEGRO sensitivity profiles is uncovered (*G*) by this experiment.



a) b)

Figure 7: Integral indices and coverage plots

This is mainly caused by different construction materials and coolants (helium vs sodium) used in adopted models resulting to dissimilar neutron spectra. The short characteristics of other identified experiments are shown in the following Table 2.

From Table 2 we can conclude that the majority of identified experiments, except MIX-COMP-FAST-001-001, are simple plutonium metal systems. The average fission group energy in these systems is quite high due to the absence of moderator and structural materials (over 1 MeV). Due to their simplicity, the *G* values get very low for all cases. The values of *g* indices for nuclide – reaction pairs, having a great impact on the neutron balance of the active core, are given in Table 3. The data presented in Table 3 highlight those nuclide – reaction pairs which are not sufficiently covered by the MIX-COMP-FAST-001-001 experiment.

Table 2: Integral indices for similar experiments in relation to ALLEGRO MOX

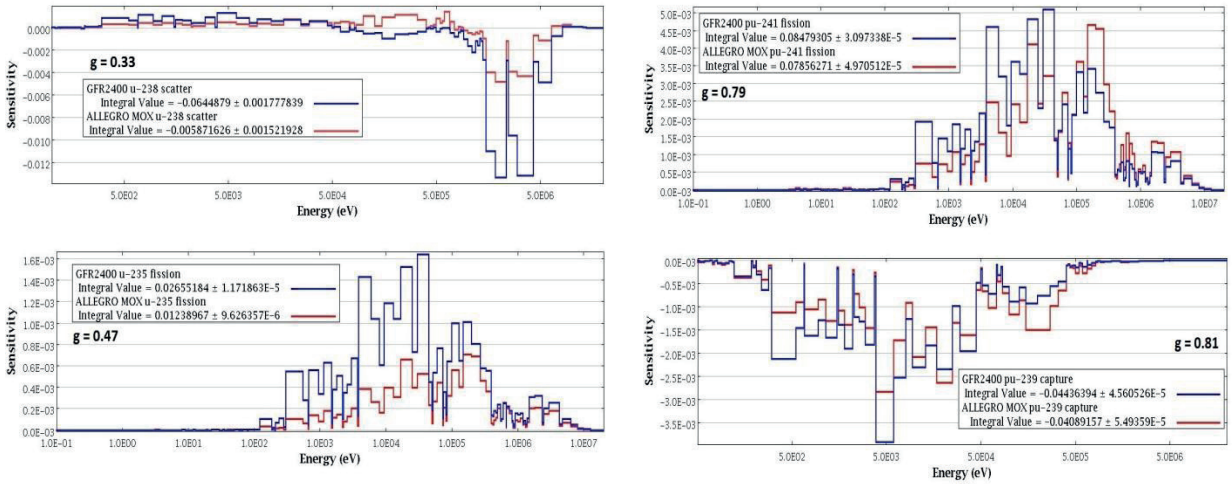
ID	ICSBEP ID	Fissile material	Moderator	Average Fission Group Energy	Neutron Flux > 100 keV	Capture > 100 keV	Cladding	Reflector	*ck*	*E*	*G*
171	MIX-COMP-FAST-001-001	MOX	Na	99.8 keV	57%	22%	SS	Depl. U	0.93	0.95	0.75
197	PU-MET-FAST-008-001	Pu Metal	-	1.08 MeV	95%	80%	-	Th	0.60	0.68	0.30
194	PU-MET-FAST-002-001	Pu Metal	-	1.28 MeV	97%	85%	-	-	0.60	0.62	0.30
199	PU-MET-FAST-018-001	Pu Metal	-	913 keV	92%	57%	-	Be	0.57	0.67	0.30
201	PU-MET-FAST-023-001	Pu Metal	-	1.17 MeV	97%	83%	-	Gr	0.56	0.63	0.26
202	PU-MET-FAST-024-001	Pu Metal	-	647 keV	95%	45%	-	PE	0.54	0.62	0.27
193	PU-MET-FAST-001-001	Pu Metal	-	1.28 MeV	97%	86%	-	-	0.54	0.60	0.26
200	PU-MET-FAST-022-001	Pu Metal	-	1.26 MeV	97%	86%	-	-	0.54	0.60	0.26
196	PU-MET-FAST-006-001	Pu Metal	-	1.11 MeV	94%	75%	-	Nat. U	0.47	0.77	0.42

Figure 7-b) shows the coverage of the most problematic nuclide – reaction sensitivity profiles by the use of all experiments involved in our calculations. The hashed area of sensitivity profiles highlights the importance of experimental verification of used nuclear data in energies in the interval between 100 keV and 1 MeV.

Table 3: The results of nuclide-reaction specific partial integral index g

ID	^{238}U capture	^{238}U total	^{239}Pu capture	^{238}U n,n'	^{240}Pu capture	^{56}Fe capture	^{58}Ni capture	^{238}U scatter	^{241}Pu capture	^{16}O capture	^{241}Am capture
171	1.00	0.99	0.98	0.99	0.46	0.90	0.47	0.96	0.20	1.00	0.35

Despite the different used fuel types (MOX vs UPuC) and reflector materials (AIM1 vs Zr_3Si_2) the similarity assessment of ALEGRO MOX and GFR 2400 core designs show quite high values of *ck* (0.85) and *E* (0.92) coefficients. The variations in material compositions are reflected by the low value of integral index *G* (0.65) where almost 35% of GFR 2400 sensitivity profiles are uncovered by ALLEGRO MOX core design. The fact that C and ^{28}Si isotopes are in ALLEGRO MOX core found in minimal amounts was confirmed by more detailed calculation of the partial integral indices. The coverage plots of specific nuclide – reaction pairs of other important contributors to integral index G are shown in Figure 8.



a) b)

Figure 8: Coverage plots of specific nuclide – reaction pairs

4 CONCLUSION

Sensitivity analysis of GFR 2400 and ALLEGRO MOX core was performed using two computational schemes where the first was based on the Monte Carlo method and second on the deterministic approach to determine spatial and energy flux distribution. Both systems are defined with different material compositions and hence by different mean neutron flux spectrum.

However the developed philosophy joins these two systems to the structure where one system serves to prove viability and feasibility of the second system. Based on this structure, some similarities in a response to change in basic data were expected which was not fulfilled on satisfactory level. In many cases, the shape of sensitivity profiles was consistent between both systems, but different absolute value of a magnitude of investigated sensitivity profiles produce presented discrepancies. The sensitivity profiles are able to predict the behaviour of a system during transients but in case of ALLEGRO MOX core and temperature reactivity effect, calculated data cannot be extrapolated for GFR 2400 core. Nevertheless, calculated database of sensitivity profiles is a useful clue for optimization process of ALLEGRO MOX core performance close to GFR 2400 or for development of experimental assembly construction. Appropriate combination of materials with moderation properties in ALLEGRO MOX core is a way how to form neutron flux spectrum to be more representative for GFR 2400 development. From the global point of view, higher sensitivity coefficients of GFR 2400, which was calculated almost in all cases, resulted to the higher uncertainty of k_{eff} induced by cross section data. This uncertainty for ALLEGRO was determined to 1.04% and for GFR 2400 the uncertainty reaches 1.67%. Special outcome of this analysis was definition of constraints in a usage of diffusion solution in perturbation theory for ALLEGRO MOX core. The similarity and uncertainty analysis of the ESNII+ ALLEGRO MOX core has identified specific problems and challenges in the field of neutronic calculations.

The similarity assessment identified 9 partly comparable experiments where only one reaches *ck* and *E* values over 0.9. However the Global Integral Index G remains still low (0.75) and cannot be considered as sufficient. The uncertainty analyses shoes that the main contributors to ALLEGRO k_{eff} uncertainty are ^{239}Pu nubar and ^{238}U inelastic scattering. The additional margin from uncovered sensitivities was determined to be 0.28%. The identified low number of similar experiments prevents the use of advanced XS adjustment and bias estimation methods. It can be concluded that more experimental data are needed. The presented results may serve as a basic step in the development of the necessary critical assemblies. Although exact data are not presented in the paper, faster 44 energy group calculation gives almost the same results in similarity analysis in comparison to more complex 238 group calculation.

The results of the similarity assessment of ALLEGRO MOX and GFR 2400 cores confirmed the adequacy of ALLEGRO MOX starting core design for the development of the commercial gas cooled fast reactor design. The real operation of ALLEGRO MOX demonstrator unit may bring new specific knowledge to the field of fast neutron reactor physics and computational methods.

Finally, it was demonstrated that TSUNAMI-IP utility can play a significant role in the future fast reactor development in Slovakia and in the Visegrad region. Clearly a further R&D and strong effort should be carried out in order to receive more complex methodology consisting of more plausible covariance data and related quantities.

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